EFFECTS OF THE DEGREE OF LAMINATION ON THE EFFICIENCY OF RADIATION SHIELDING

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by

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I. INTRODUCTION

A. Statement of Problem

Fundamental knowledge of the interactions of neutrons and gamma radiation with matter is essential for an understanding of nuclear shielding. Since the purpose of any radiation shield is to protect personnel and equipment, the important neutron reactions in nuclear shielding are those which lead to a reduction of the neutron flux. The microscopic neutron cross sections have been determined for the various elements (3), and it can be seen from these cross sections that the probability of capturing a slow neutron is greater for most elements than is the probability of capturing a fast neutron. Since neutrons are produced in the fission process at relatively high energies, they must be slowed down before they can be removed from the system by being captured. Gamma rays are produced when a neutron is absorbed by the nucleus of most elements. It is the function of a radiation shield to slow down the fast neutrons to thermal energies, to remove these thermal neutrons by the capture process, and to protect personnel and equipment from the gamma radiation of the source and the additional gamma radiation which is produced by the neutron capture process.

Most shielding studies have been concerned with the large permanently located reactors, and concrete has been extensively used to fulfill the shielding requirements of these reactors.

If nuclear power plants are to be mobile, it is apparent that the present shielding ideas must be expanded and should include other concepts of shielding materials which could be used in the place of concrete to reduce both the size and the weight of the radiation shield. When materials other than concrete are considered for use in radiation shields, the problem of the effects of their geometrical arrangement within the shield becomes important.

In this investigation, an attempt was made to use materials which by themselves would have fulfilled only one of the functions of a radiation shield. The radium-beryllium neutron source which was used in this investigation produced radiation similar to the radiation encountered in shielding a nuclear reactor. The object of this investigation was to compare the relative effectiveness of these materials for shielding against this type of radiation as a function of their geometrical arrangement within the shield.

The practical results of such a study will assist in determining the best type of shield construction.

B. Review of Literature

The literature review did not disclose any previous investigations closely related to the effects of the degree of lamination on the efficiency of radiation shielding.

Many investigations have been conducted in the field of radiation shielding, but in most cases they have been made to determine the effects of the composition of the concrete and the effects of variously shaped ducts through the shield.

C. Description of Experiment

A cycle of shielding material is defined in this report to be paraffin, cadmium, and lead, in this order. The paraffin was used as the moderating material with the function of slowing the neutrons down to thermal energies, the purpose of the cadmium was to capture the thermal neutrons, and the principal function of the lead was to serve as a shield for both the gamma radiation of the source and the gamma radiation produced by the neutron-gamma reaction in the cadmium. The shielding materials used in this investigation consisted of ten ½-in. sheets of paraffin, ten 0.001-in. sheets of cadmium, and ten ¼-in. sheets of lead. Since each configuration of the shield contained the same ten pieces of each type of material, the minimum number of cycles of shielding

materials was one, while the maximum number was ten. In the one-cycle shield, all ten sheets of paraffin were placed next to the source, then the ten sheets of cadmium foil, and then the ten sheets of lead. The ten-cycle configuration of the shield was made by using alternate single sheets of paraffin, cadmium, and lead.

With all other variables held constant, the number of cycles of shielding materials was varied from one to ten, and measurements were made of the slow neutron flux, the fast neutron flux, and the intensity of the gamma radiation for each configuration. Since it was desired to have equal amounts of each type of material in every cycle of any one configuration, there were only the one-cycle, two-cycle, five-cycle, and ten-cycle configurations possible with ten pieces of each type of shielding material. No attempt was made to determine the absolute value of the neutron flux or the absolute intensity of the gamma radiation at the surface of the shield, since the counts per minute and the roentgens per hour recorded gave direct indications of the relative efficiency of the shields.

II. MATERIALS AND APPARATUS

A. Shielding Materials

Paraffin was chosen for the moderating material for this investigation because of its availability, low cost, ease of use in the formation of the various configurations of the shield, and its relatively high moderating ratio. The moderating ratio is one of the most important quantities, from the theoretical standpoint, in expressing the effectiveness of a moderator (2). Some approximate values of this ratio, using thermal absorption cross sections, were computed by the formula below and are listed in Table 1.

$$\text{Moderating Ratio} = \frac{\Sigma_s \xi}{\Sigma_A} \qquad (2)$$

where:

Σ_s is the macroscopic scattering cross section.

Σ_A is the macroscopic absorption cross section.

ξ is the average logarithmic energy decrement per collision.

A is the atomic weight.

$$\xi \approx \frac{2}{A + 2/3} \qquad (2)$$

The mean value of the average logarithmic energy

Table 1

Moderating Ratio of Materials

Material	Moderating Ratio
Water	162.0
Paraffin	131.0
Beryllium	160.0
Carbon	169.0
Lead	0.4

decrement per collision $\bar{\xi}$, which should be used in the above formula for neutrons slowing down in a system of several nuclear species, is defined by

$$\bar{\xi} \equiv \frac{\sum_{i=1}^{N} \Sigma_{si} \xi_i}{\sum_{i=1}^{N} \Sigma_{s}} = \frac{\sum_{i=1}^{N} \Sigma_{si} \xi_i}{\Sigma_s} \qquad (2).$$

The microscopic cross section, for the hydrogen in water and in paraffin, used in the calculations of the moderating ratios listed in Table 1, was taken for the case where the atom is bound in a molecule with the thermal neutron energy of 0.025 electron volts (2).

The paraffin which was used in this investigation was the type which is normally used for molding candles. Since the exact molecular formula for this paraffin was not known,

it was assumed in the above calculations that it was the normal C_nH_{2n+2} compound (1) with n equal to 26. This paraffin was purchased from a local concern in ten-pound blocks which were approximately 1½ by 11 by 19 in. These blocks were cut to the desired dimensions (¼ by 8 by 16 in.) by a band saw. Due to the difficulty experienced in trying to saw eight-inch widths, the paraffin was cut into pieces ¼ by 4 by 16 in., and these pieces were welded together by a hot iron to form the desired sheets. This method of cutting the paraffin gave some variations in thickness, but it is believed that this source of error was eliminated by numbering the sheets and arranging them in the same order for each configuration of the shield.

The cadmium used for neutron absorption was purchased from the Division Lead Company, Chicago, Illinois, in 0.001 by 8 by 16 in. sheets. The conventional calculations indicated that ten of these sheets of cadmium foil would capture approximately 99 per cent of the thermal neutrons.

The lead was purchased from the National Lead Company, St. Louis, Missouri in ¼ by 8 by 16 in. sheets. No computations were made to determine the thickness of lead required to reduce the gamma radiation to some desired level, and the thickness of the lead was arbitrarily chosen to be the same as the paraffin.

B. Detecting Apparatus

A standard nine-inch B^{10} lined neutron counting tube with a preamplifier and a standard scaling circuit (Figures 1 and 2) was used for the detection of neutrons. The preamplifier (Figure 3) was designed by the electronic shop of the Atomic Energy Commission at Iowa State College, but some scaling circuits have built-in amplifiers which work equally well. The operating characteristics of this equipment (Appendix A) were found to be satisfactory, but the narrow plateau (Figure 4) required a regulated voltage supply for accurate results.

A General Electric Radiation Monitor (Figure 1) and a standard Geiger-Müller tube were used for the detection of gamma radiation. Experiments were performed (Appendix B) to determine the operating characteristics of these detectors, and they were found to be satisfactory.

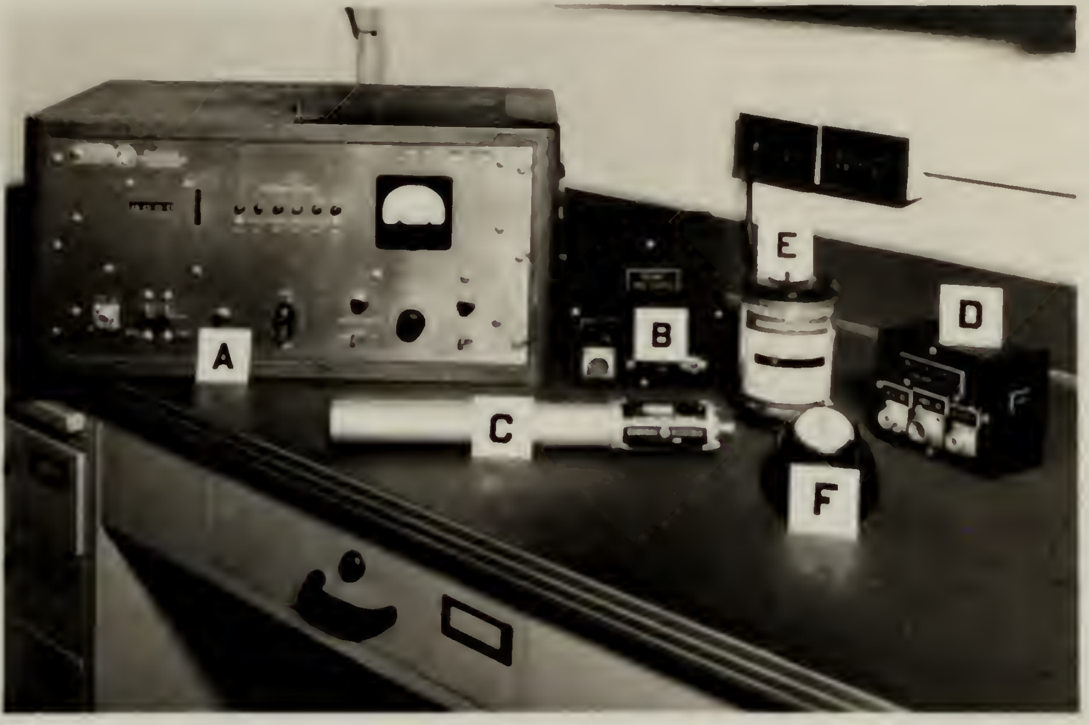
C. Source and Accessory Apparatus

A radium-beryllium neutron source which contained 94.7 mg. of radium was used as the source of radiation. This source was left in its container with the lead plug removed from the access hole (Figure 7) for this investigation.

A wooden platform was constructed to fit over the lead

Figure 1. Detecting Apparatus

A -- Scaler
B -- Preamplifier power supply
C -- B^{10} lined neutron counting tube
D -- Preamplifier for B^{10} tube
E -- General Electric Radiation Monitor
F -- Stop watch


A
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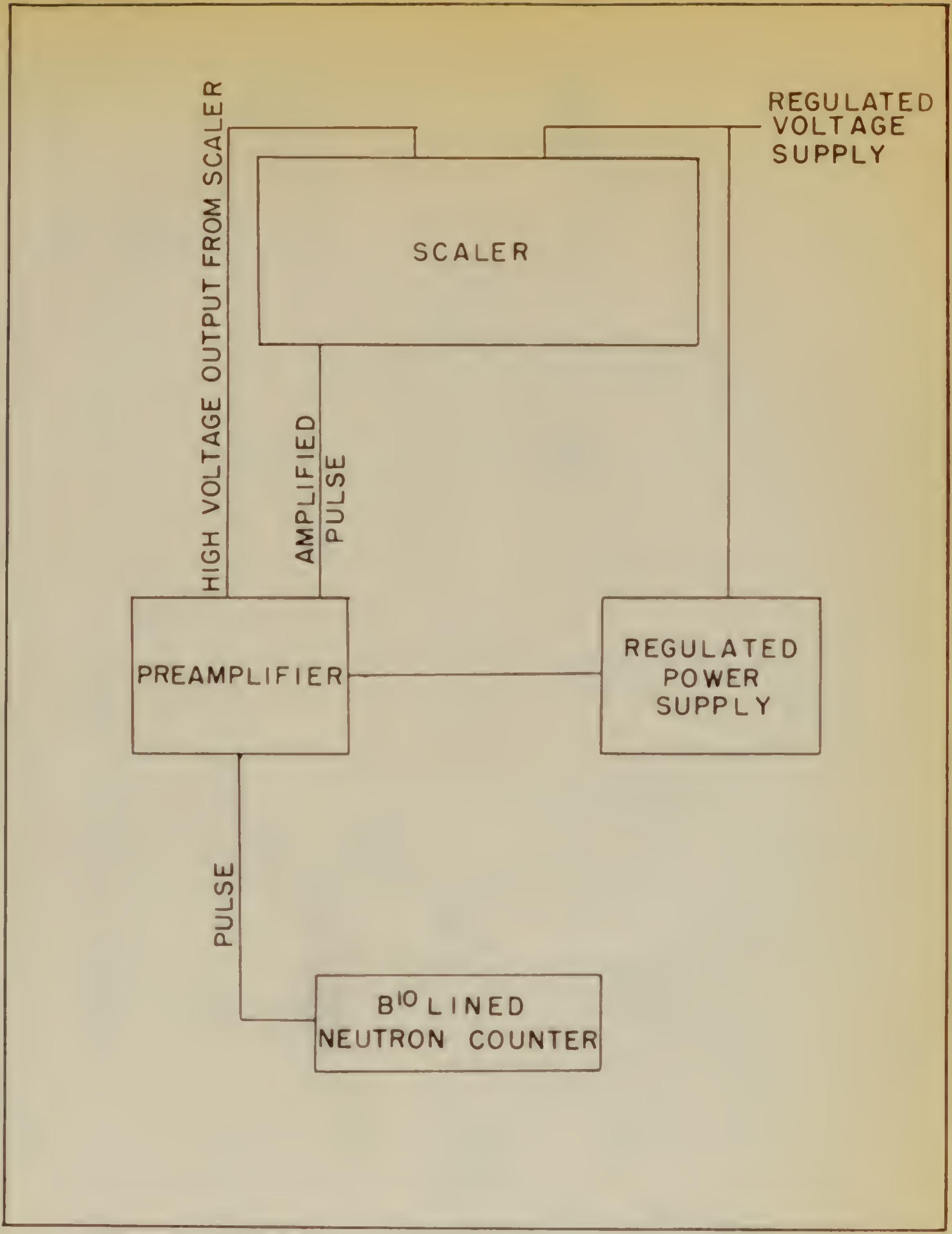

Figure 2. Neutron Counting Circuit -- Block Diagram.

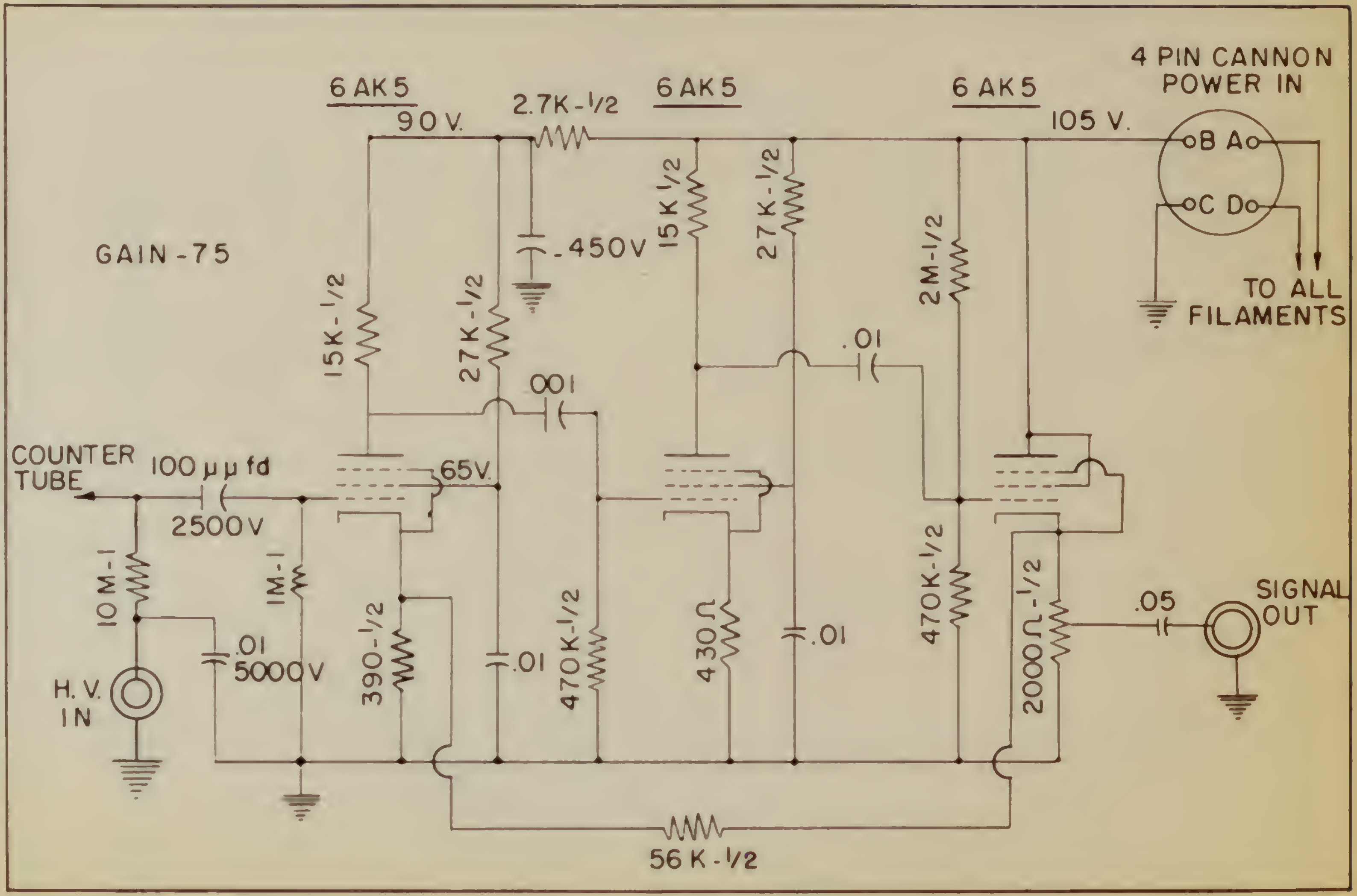

Figure 3. Preamplifier Circuit for B^{10} Lined Counter.

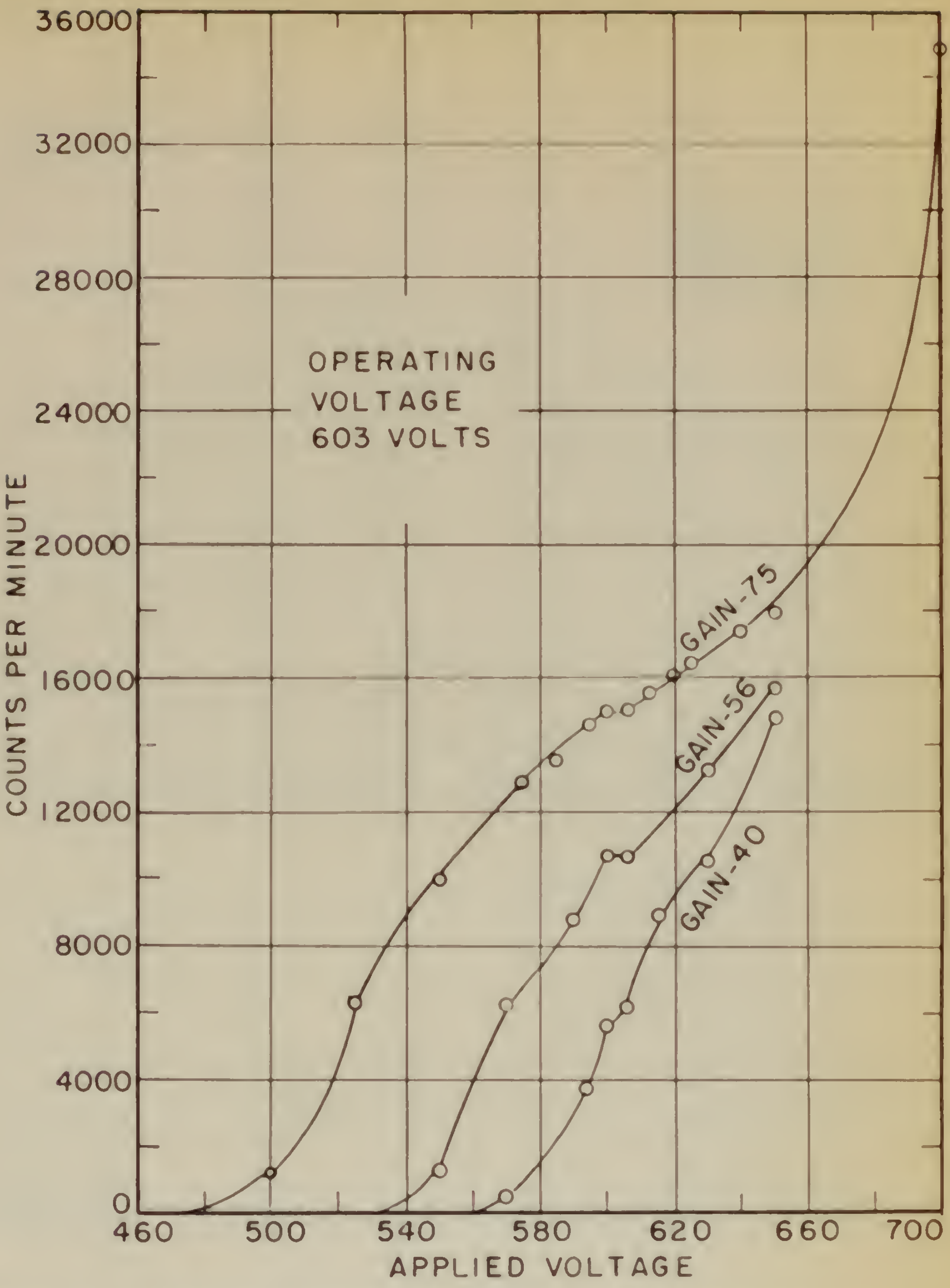

Figure 4. Plateau Curves for B^{10} Lined Neutron Counter.

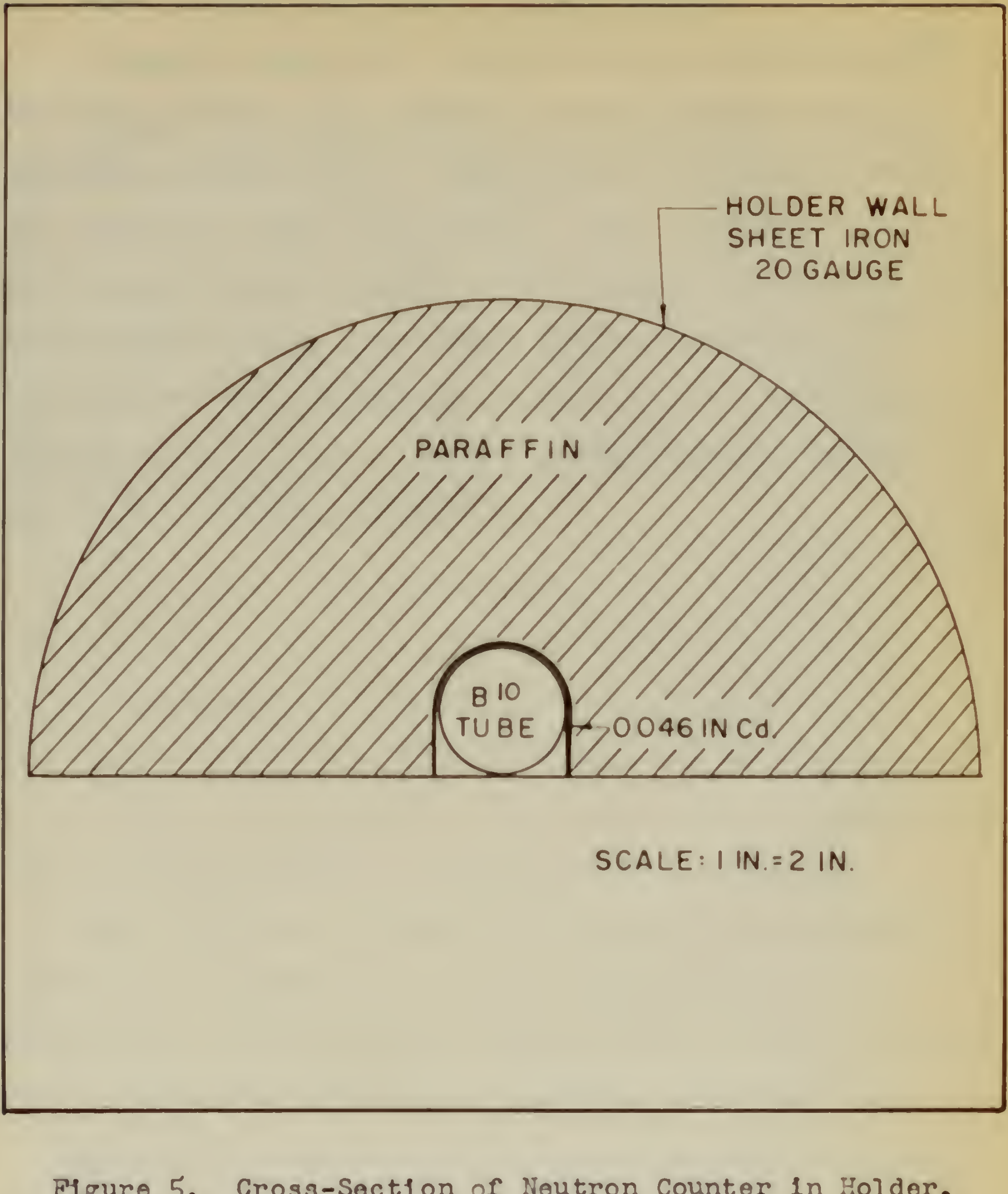

Figure 5. Cross-Section of Neutron Counter in Holder.

ball of the source container to hold the shielding materials in a fixed position above the neutron source.

A detector holder was constructed to position the B^{10} lined tube at the same position for each configuration of shielding materials and to shield it from neutrons that had been scattered around the shield. This shielding of the detector from scattered neutrons was accomplished by filling the holder with paraffin and placing cadmium around the B^{10} lined counter tube (Figure 5) so that the only neutrons entering it would have to come directly from the source through the shielding materials.

III. PROCEDURE

Figure 6 is a photograph of the experimental arrangement, showing the neutron counter in its holder, which was positioned over the shielding materials as shown in Figure 7. The counting time for the various configurations and arrangements (Figure 7) was varied to maintain a statistical accuracy of approximately one per cent. All of the neutron data were taken the same day to avoid any possible error being induced due to changes in the operating characteristics of the equipment. The neutron source was maintained in a fixed location in the room during the experiment to insure that the neutron flux within the detector holder, resulting from neutrons being scattered around the shielding material by the walls and equipment in the room, did not change due to changes in the geometry of the scattering materials.

Since the neutron source was enclosed in a seven-inch diameter lead sphere with a hole as shown in Figure 7, removing the lead plug which normally fits in this hole produced a semi-collimated beam of neutrons and gamma rays.

The shield was centered over the hole, with the first sheet of shielding material even with the top of the lead ball. With the shield in place, the detector holder was placed over the shield, and was positioned by wooden strips

Figure 6. Experimental Set-up.

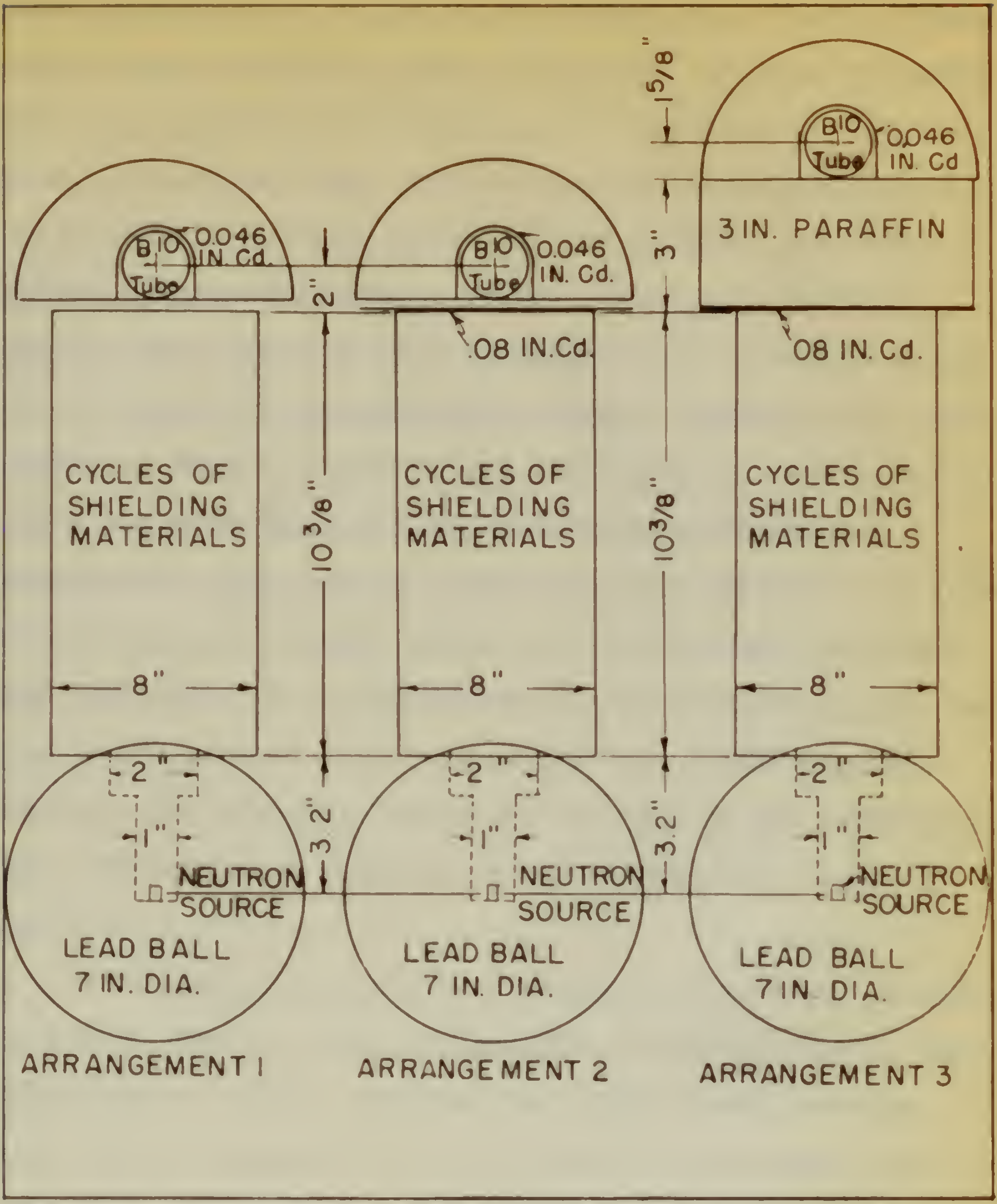

Figure 7. Experimental Arrangements for Neutron Counting.

which were attached to the source container. The B^{10} lined counter tube was fixed within the holder, and was not moved during the entire investigation. The small space between the shield and the detector holder in arrangement 1 (Figure 7) was to make it possible to insert the 0.08-in. sheet of cadmium between the shield and the detector holder in changing from arrangement 1 to arrangement 2 without moving any portion of the experimental set-up. Although the reproduction of data by removing and replacing the detector holder was good (Appendix A), this method of changing arrangements gave complete assurance that no errors were induced by changes in the geometry of the set-up. In going from arrangement 2 to arrangement 3, the detector holder was removed and three inches of paraffin was placed over the cadmium. The detector holder was aligned in the horizontal plane by the wooden strips, and rested directly on the paraffin.

By forming the particular configuration of the shield, and taking readings with arrangement 1, arrangement 2, and arrangement 3 without changing the shield configuration, errors due to changes in the geometry of the set-up were reduced to a minimum.

The General Electric Radiation Monitor and the Geiger-Muller tube were positioned over the shield by a wooden frame, and since long irradiation times were necessary for

the General Electric Radiation Monitor, this frame was nailed in position to insure that there were no changes in the geometry of the set-up during the measurement.

The counting rate with the Geiger-Müller tube was sufficiently high that ten-minute counts gave the desired one per cent statistical accuracy. The investigation using the Geiger-Müller tube was performed in one day with the background observed after the neutron source was removed from the room. Since no attempt was made to isolate the Geiger-Müller tube from the radiation scattered around the shield, the observed background was not the true value, but it is believed that the additional background due to scattered radiation was constant and that this did not induce any additional errors in the results of this investigation.

The gamma radiation intensity at the surface of the shield was found to be approximately 0.4 mr/hr. Since the General Electric Radiation Monitor has a full scale deflection of twenty milliroentgens, an irradiation time of approximately thirty hours was necessary to give half-scale deflection.

IV. RESULTS AND DISCUSSION

A. Neutron Counts

After obtaining the neutron counts listed in Table 2 by using the three arrangements shown in Figure 7, it was possible to separate these counts into those due to fast neutrons and those due to slow neutrons. It can easily be shown that the probability of a thermal neutron penetrating 0.08 in. of cadmium is essentially zero. Since in arrangement 1 the neutron counts recorded were from neutrons of all energies and in arrangement 2 essentially all the neutrons with energies below 0.2 ev. had been removed by the cadmium, the difference between these two counts was the counts due to slow neutrons, which are defined here to be neutrons with energies less than 0.2 ev.

Since the B^{10} lined counter is much less sensitive to fast neutrons than it is to slow neutrons, arrangement 3 was used to slow down the fast neutrons which passed through the cadmium in order to increase the counting rate for the fast neutrons. This higher counting rate decreased the effects of the scattered neutrons which may have leaked through the cadmium shield within the detector holder. Since the detector in arrangement 3 was at a greater distance from the source

Table 2

Neutron Counts

Arrangement	Cycles of Shielding Materials	Counting Time (Minutes)	Counts	Counting Rate (R) $R = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}}$
1	1	20	22850	1142.5 ± 7.5
1	2	30	20100	670.0 ± 4.7
1	5	30	13830	461.0 ± 3.9
1	10	40	16008	400.2 ± 3.2
2	1	60	11010	183.5 ± 1.8
2	2	60	10060	167.7 ± 1.7
2	5	60	9186	153.1 ± 1.6
2	10	60	9030	150.5 ± 1.6
3	1	30	12126	404.3 ± 3.5
3	2	40	14030	350.7 ± 3.0
3	5	40	12836	320.9 ± 2.8
3	10	40	12162	303.0 ± 2.8

than in the other two arrangements, the neutron counts recorded were corrected by the ratio of the squares of the distances between the source and the detector. This ratio was computed from the distances shown in Figure 7, and its value was found to be 1.47.

The above methods were used for the interpretation and correction of the neutron counts in Table 2, and the counting rates for slow and fast neutrons are listed in Table 3 and plotted in Figures 8 and 9. It can be seen from these plots that the degree of lamination of the shielding materials has a pronounced effect on the efficiency of a radiation shield.

Table 3

Slow and Fast Neutron Counts

Cycles of Shielding Materials	Slow Neutron Counting Rate $(R_1 - R_2 \pm \sqrt{\sigma_1^2 + \sigma_2^2})$*	Fast Neutron Counting Rate $(1.47 \times R_3)$*
1	959.0 ± 7.7	592.0 ± 5.1
2	502.3 ± 5.0	515.0 ± 4.3
5	307.9 ± 4.2	471.0 ± 4.1
10	249.7 ± 3.6	446.0 ± 4.1

*Subscripts refer to the arrangements in Figure 7.

$$\sigma = \frac{\sqrt{\text{Counts}}}{\text{Time in Minutes}}$$

The manner by which these curves approached the value for the ten-cycle configuration indicated that the degree of lamination of a radiation shield could be increased to a point at which a laminated shield would have the same efficiency as a homogeneous mixture of the shielding materials. From some tests performed (Appendix D) to check this indication, it was found that there was fair agreement between the ten-cycle shield and the theoretical behavior of a homogeneous shield (Figure 14). These results were somewhat irregular, but it is believed that this could have been caused by variations in the thickness of the individual cycles and the variation in the distance between the shield and the detector as the cycles were added.

Figures 8 and 9 show a decrease in both the slow neutron

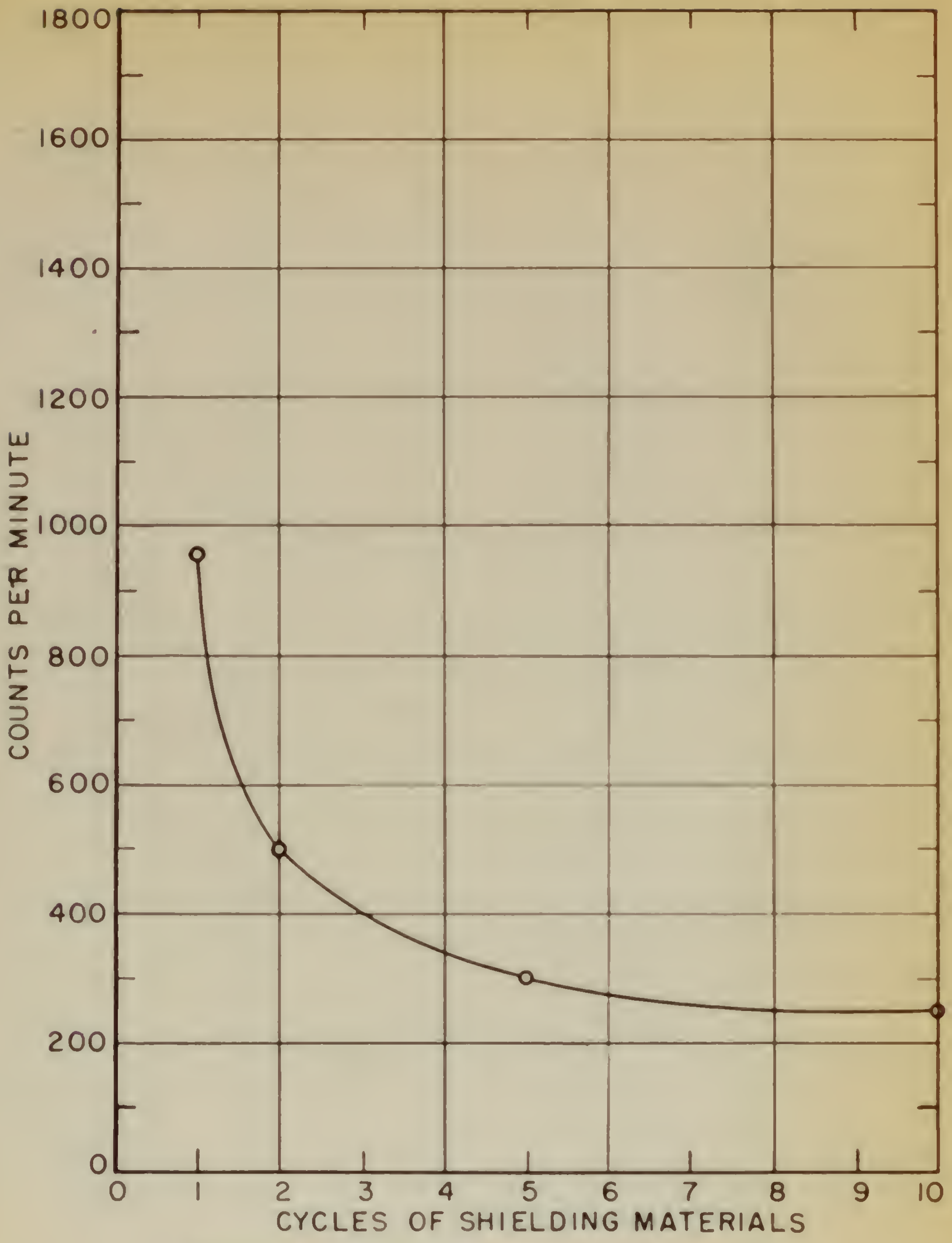

Figure 8. Slow Neutron Count.

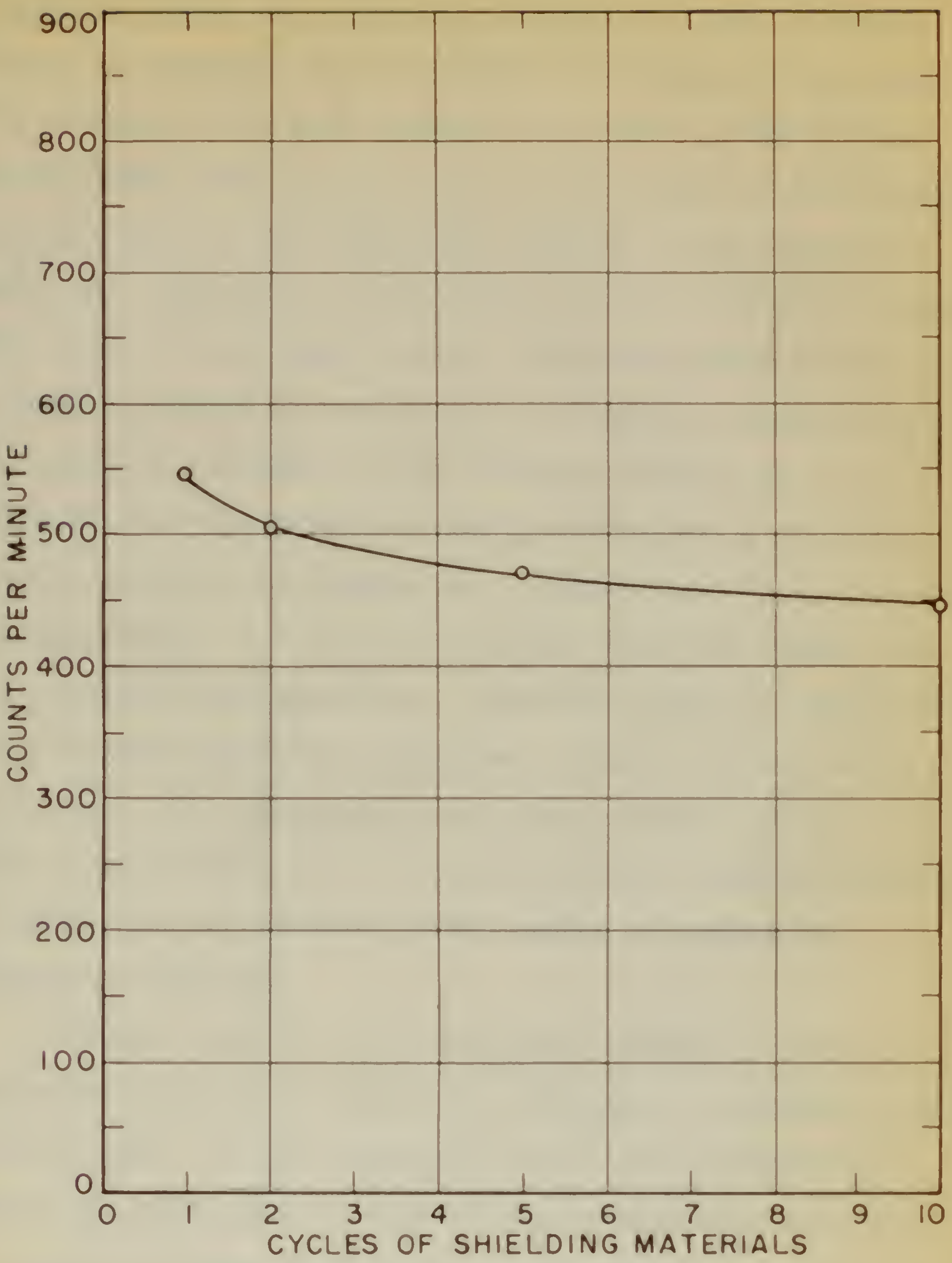

Figure 9. Fast Neutron Count.

count and the fast neutron count for an increase in the number of cycles of shielding materials. Since the same number of neutrons started through the shield in each configuration, either more neutrons were lost by scattering through the edges of the shield or the degree of lamination had an effect on the moderating ability of the materials. Since the difference in the losses due to scattering through the edges of the shield for the different configurations was checked (Appendix C) and found to be small in comparison to the observed decrease in slow neutrons (Figure 8), it is believed that the ten-cycle configuration had a slightly greater ability for slowing down neutrons and consequently a greater ability for removing neutrons from the system than the one-cycle configuration. Figure 10 shows the microscopic cross section curves for hydrogen, carbon, and lead which were taken from Reference 3, and these curves will be used to give an explanation for the experimentally observed decrease in counts for an increase in the number of cycles of shielding materials.

Although lead is a poor moderator (Table 1) there is a finite probability of a neutron undergoing a scattering collision in lead. In the one-cycle shield, the paraffin performed the moderation of the neutrons which were removed by the cadmium without the effects of the lead. Some of the

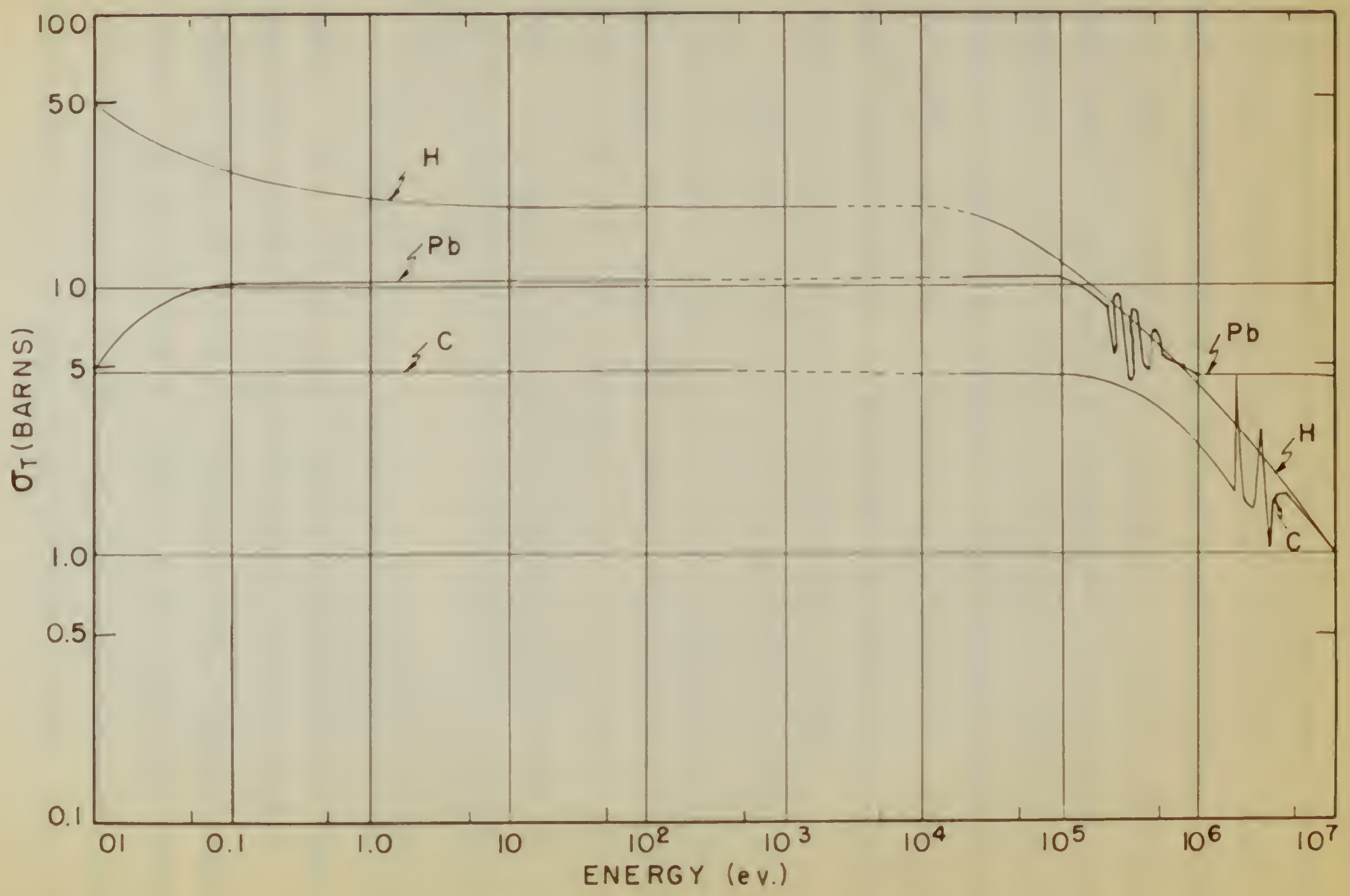

Figure 10. Neutron Cross Sections.

neutrons which passed through the cadmium, experienced scattering collisions in the lead and were counted as slow neutrons. In the ten-cycle configuration of the shield, any slow neutrons produced by the moderation occurring in the lead could be removed from the system. From the analysis of the curves in Figure 10, it is obvious that any moderation occurring in the lead at any energy level cannot adversely affect the moderating ability of the paraffin, but if this moderation occurs in the energy range where the microscopic cross section curves for hydrogen and carbon have negative slopes it will improve their moderating ability.

This increase in the moderating ability of the paraffin in the ten-cycle configuration by the moderating effects of the lead, and the slightly greater losses due to scattering, explain the observed decrease in the fast neutron count. These effects, plus the more efficient positioning of the cadmium, explain the observed decrease in the slow neutron count.

The results of this investigation (Figures 8 and 9) indicate that the ten-cycle configuration of these three shielding materials is more efficient for attenuating neutrons than the one-cycle configuration. Before these results can be extended to include other shielding materials, the magnitude of the increase in the moderating ability of the paraffin by the effects of the lead, and the magnitude

of the increase in losses due to the neutron scattering through the edges of the shielding materials should be investigated and compared to the over-all increase in effectiveness of the shielding materials. The magnitude of the increase in moderating effectiveness is governed by the shape of the microscopic cross section curves of the shielding materials, and the effects of losses due to scattering can be reduced by the use of larger sheets of shielding materials.

B. Gamma Counts

The gamma counts as recorded by the Geiger-Müller tube are listed in Table 4 and plotted in Figure 11. The net counting rate has been corrected for the observed background count of 36.1 ± 0.6.

Table 4

Gamma Counts by Geiger-Müller Tube

Cycles of Shielding Materials	Counting Time (Minutes)	Counts	Net Counting Rate (R_n) $R_n = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}} - \text{Bk.}$
1	10	11324	1096.3 ± 10.6
2	10	10568	1020.7 ± 10.3
5	10	10260	989.9 ± 10.2
10	10	10256	989.5 ± 10.2

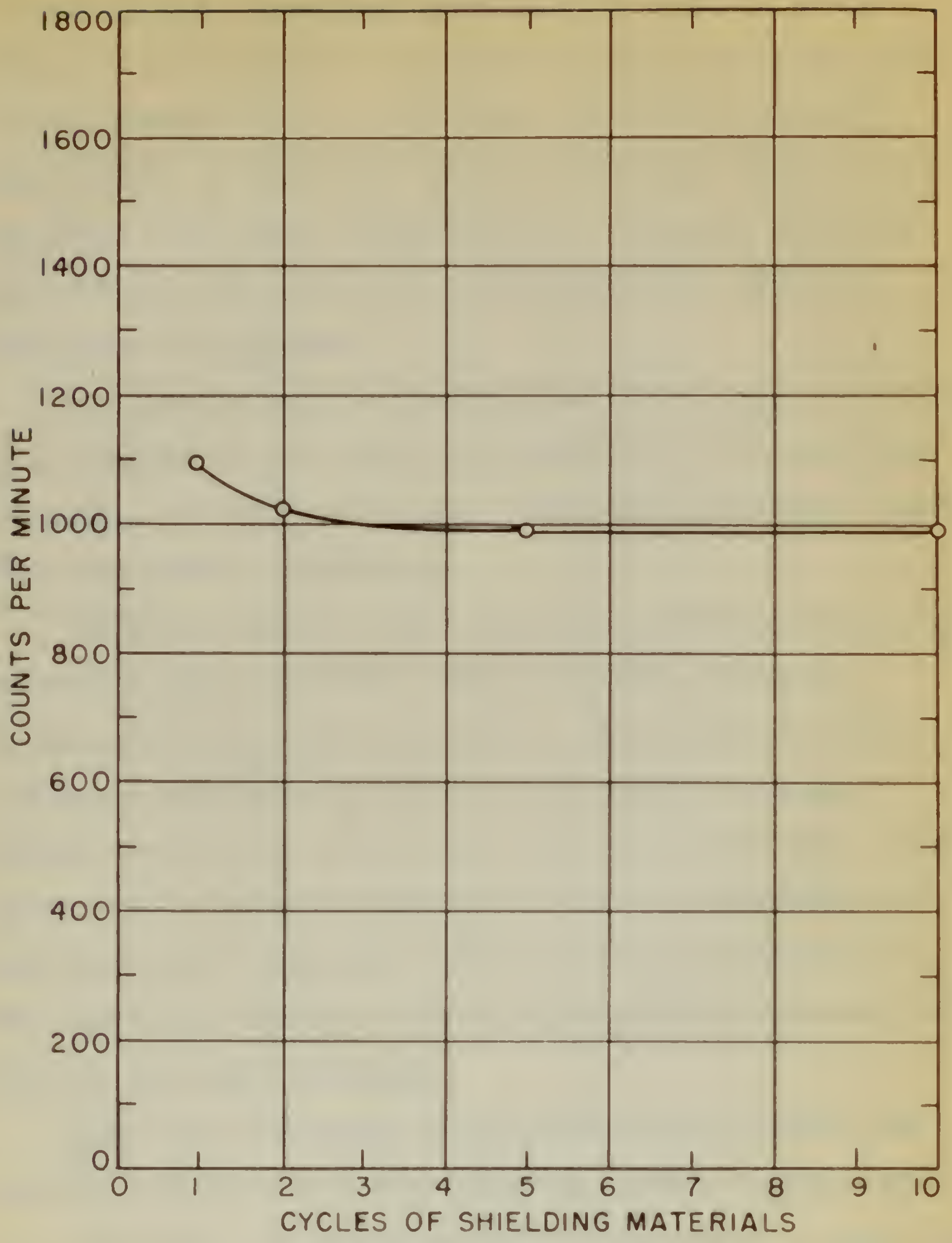

Figure 11. Gamma Count.

Table 5 is the gamma intensity as recorded by the General Electric Radiation Monitor. The probable error listed in this table is the combined error due to both the assumed reading error and the experimentally determined reproduction of data error of the instrument (Appendix B). The gamma count curve (Figure 11) and the gamma intensity curve (Figure 12) both show a decrease as the degree of lamination is increased.

In order to explain the behavior of the gamma radiation as a function of the degree of lamination of the shielding materials, the three methods by which gamma radiation reacts with matter will be reviewed.

The photoelectric effect and pair production completely remove the gamma radiation from the system, while the Compton effect degrades its energy. The curves for mass absorption coefficients versus gamma energy, with and without the Compton effect, have been determined (6). These curves show a definite increase in the mass absorption coefficients with decreasing energies below approximately 2 Mev. for heavy elements such as lead and below approximately 0.1 Mev. for light elements.

Since the efficiency of the Ra-Be neutron source for producing neutrons from the available alphas is very small, and since there are approximately as many gammas as there are

Table 5

Gamma Intensity

Cycles of Shielding Materials	Irradiation Time (Hours)	Reading (mr)	Intensity (mr/hr.)	Probable Error
1	24.75	9.8	0.396	±0.0047
2	25.20	9.7	0.385	±0.0046
5	38.67	13.9	0.360	±0.0039
10	31.00	11.2	0.362	±0.0040

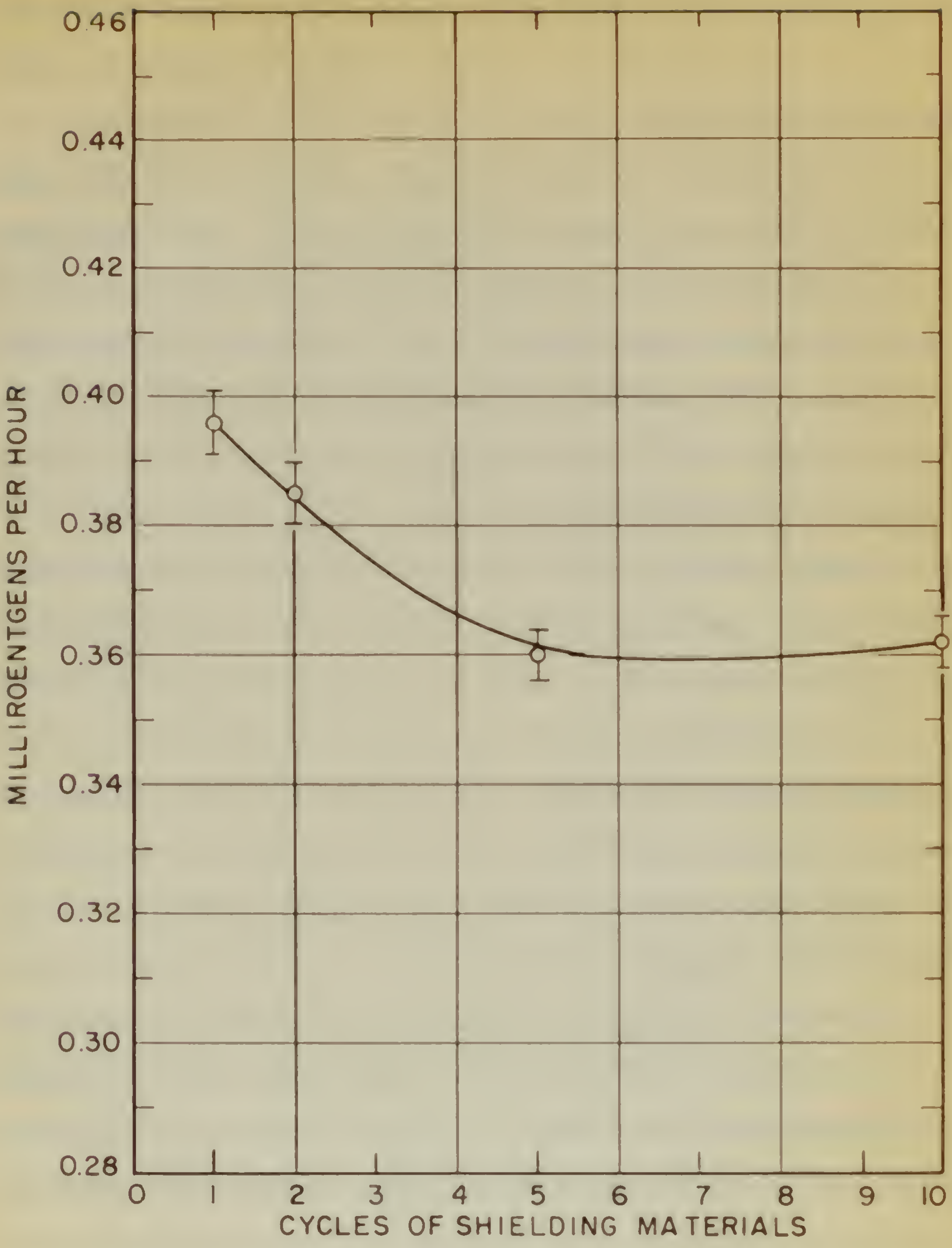

Figure 12. Gamma Intensity.

alphas in the radium decay products (4), it is assumed that the contribution of the gammas produced in the cadmium to the total intensity is small.

A comparison will be made of the penetration of the one-cycle and the ten-cycle configuration of the shield by a 1.8 Mev. gamma ray. In the one-cycle shield, the entire effect of the paraffin for shielding against gamma radiation of this energy is small and the effective mass absorption coefficient will have an average value within this energy range. In the ten-cycle configuration of the shield, the first $\frac{1}{2}$ in. of lead will reduce the intensity of the gamma radiation to approximately one-half its original value, and successive layers will continue this reduction. Since this reduction of gamma intensity occurs in the energy range where there is a finite probability of Compton scattering, some of the gamma rays will undergo this effect and will be reduced in energy. If the assumption is made that paraffin follows the same trend as carbon, air, water, tissue, and oxygen (6), it is obvious that as the lead in the ten-cycle configuration degrades the gammas to energies below 0.1 Mev. the mass absorption coefficient for the photoelectric effect in paraffin is greatly increased. Since the effectiveness of paraffin for shielding against gamma radiation is very small in comparison to lead, it is logical to assume that the 5 in.

of paraffin in the one-cycle configuration of the shield will have very little effect on degrading the energy of the gammas, and consequently very little effect on the mass absorption coefficient for lead. Since the magnitude of the increase in the mass absorption coefficient for the photoelectric effect for paraffin in the ten-cycle configuration is an appreciable amount, it follows that there should be a net gain in the ability of the materials in the ten-cycle configuration to shield against the gamma radiation of the source.

The results of this investigation indicate that the materials in the ten-cycle configuration of the shield are more effective in shielding against gamma radiation than they are in the one-cycle configuration.

The foregoing analysis of the curves of the mass absorption coefficients versus energy (6) explains the decrease in the intensity of the gamma radiation with increased lamination, if the assumption is made that the gammas produced in the cadmium are only a small fraction of the gamma radiation in the system. Although this is considered to be a valid assumption for this investigation, it will not necessarily be valid for other shielding problems. For this reason, these results cannot be extended to other shielding materials or other sources of radiation without a complete analysis of the system.

C. Supplementary Neutron Counts

The lower-half of the lead ball shown in Figure 7 was surrounded by paraffin, and it was believed that this could have been an additional factor which was not considered in the foregoing discussion. There is a finite probability that some of the neutrons that enter this paraffin will be scattered through its upper surface at angles such that they will enter the shield through the edges of the shielding materials, and there will be a difference in the counts recorded due to those scattered neutrons. This difference will be due to the geometrical arrangement of the cadmium in the shield, which will capture a greater percentage of these neutrons in the ten-cycle configuration than in the one-cycle configuration.

An additional experiment was performed to determine the neutron counting rate with the edges of the shielding materials protected from these scattered neutrons. The description and the results of this experiment are listed in Appendix E.

The available data are not adequate to allow a complete interpretation of these results. The total count, as plotted in Figure 15, indicates that a maximum efficiency occurs with the two-cycle configuration. If the previously obtained fast neutron counts were subtracted from the total count to obtain the slow neutron count, these results would indicate that the

one-cycle configuration was the best with the minimum efficiency occurring at the five-cycle configuration. If the differences in the scattering losses were included in either of the two preceding interpretations, the results would indicate that the maximum efficiency occurs at the two-cycle configuration and that there is very little difference in the efficiency of the five- and the ten-cycle configurations. It can be seen from these three possible interpretations that the data do not indicate conclusively the variation in efficiency of shielding as a function of lamination for the case in which the edges of the shielding materials were enclosed in cadmium.

V. CONCLUSIONS

The following conclusions appear to be justified for the shielding materials and source of radiation which were used in this investigation.

1. For gamma radiation, fast neutrons, and slow neutrons, the ten-cycle configuration of the shield was more effective than the one-cycle configuration when the edges of the shielding materials were not enclosed by cadmium.

2. The moderating ability of the shield increased an undetermined amount as the degree of lamination was increased when the edges of the shielding materials were not enclosed by cadmium.

3. The degree of lamination had little effect on the efficiency of the shield for the attenuation of neutrons with the edges of the shielding materials enclosed by cadmium.

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Since my work at Iowa State College was the third and final year of the Aeronautical Engineering Curriculum of the United States Naval Postgraduate School, Monterey, California, I should also like to express my appreciation to the Naval Postgraduate School for making my work at Iowa State College possible, and to Dr. G. R. Town of the Iowa Engineering Experiment Station for making available the expendable materials used in this investigation.

VIII. APPENDICES

A. Operating Characteristics of the Neutron Detector

Preliminary investigations were made to determine the operating characteristics of the neutron detector. The information available on the counter used was not adequate to provide the operating voltage, the effects of the preamplifier gain, and the relative sensitivity of the counter to neutrons in an area of both neutrons and gamma radiation.

A series of plateau curves (Figure 4) were determined with a preamplifier gain of 40, 56, and 75. It was found that the plateau occurred at 603 volts regardless of the preamplifier gain, but that the slope of the curve in the plateau region was greater for a gain of 40 than for gains of 56 and 75. This investigation showed that the operating voltage was 603 volts, that the preamplifier gain should be between 56 and 75, and that a regulated voltage supply was required for the scaler circuit.

Since the preamplifier had a maximum gain of 75 which occurred at a control setting of 94 on a 100 division scale, the preamplifier gain was computed by the formula:

$$\text{Preamplifier Gain} = \frac{75}{94} \times \text{Control Setting}.$$

The variation in the counts recorded versus preamplifier gain (Figure 13) was investigated, and this variation was found to be linear for preamplifier gains between 40 and 70. Since the preamplifier gain was manually adjusted and remained at a particular setting throughout the investigation, the variation of gain due to changes of the gain control was not a source of error. Since the statistical accuracy is a function of the counting rate and the counting time, it was decided to use the maximum preamplifier gain of 75 to obtain the maximum counting rate.

The efficiency of the neutron counter for counting gamma radiation was found to be negligible for gammas with energies in the radium spectrum. A 10.02-mg. radium source was placed in contact with the B^{10} lined neutron counter tube and the counts recorded were the same, within statistical accuracy, as the background. Since the general shape of the curves for mass absorption coefficients versus gamma energy (6) show a decrease in the probability of a reaction occurring with gammas of energies greater than the highest energy gamma in the radium spectrum, it is believed that the above results can be assumed to hold for the gamma radiation for the neutron-cadmium reactions. Most of the gammas produced in the neutron-cadmium reaction have energies within the energy spectrum of the radium gammas, but there is some

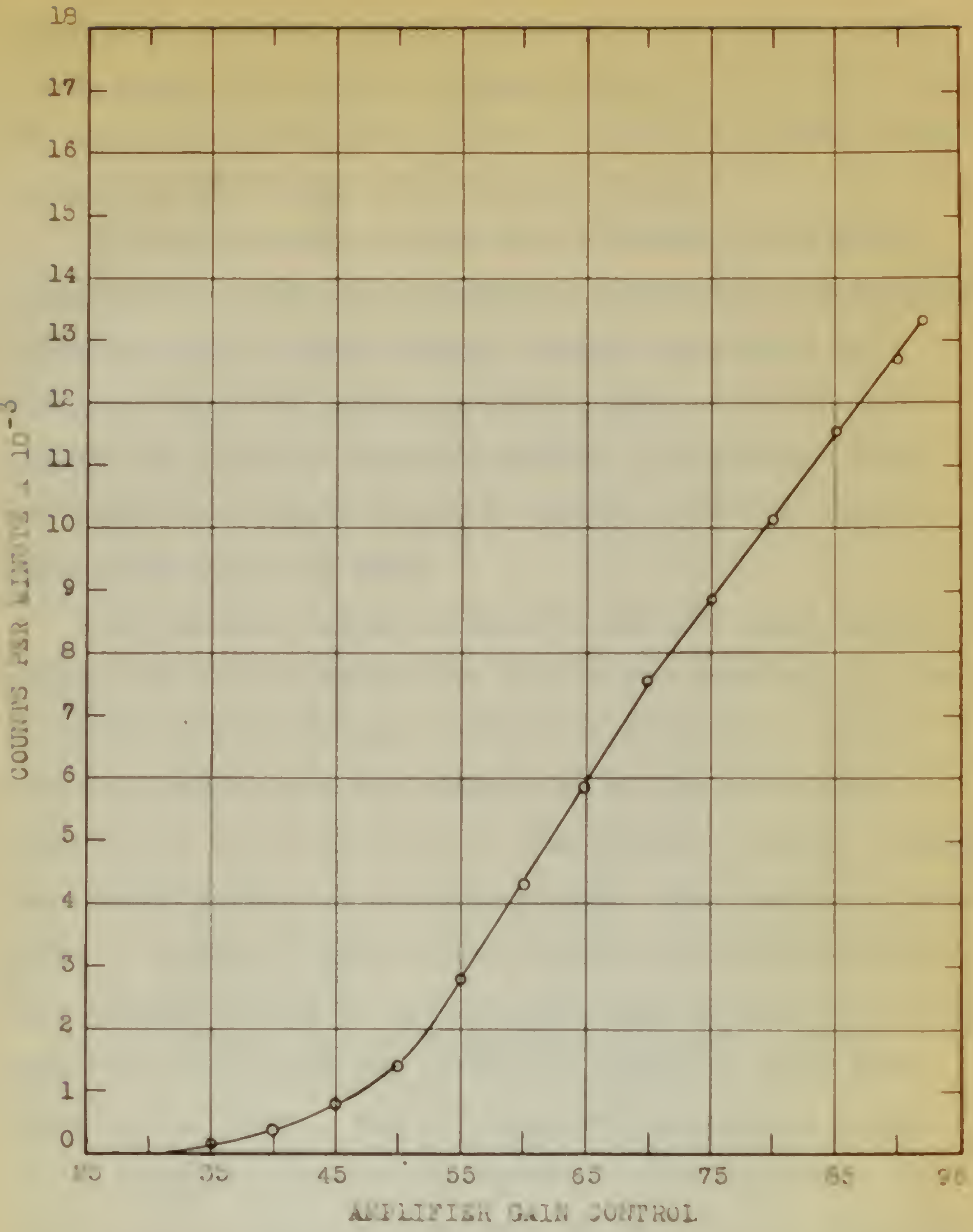

Figure 13. Counts Recorded Versus Amplifier Gain.

possibility of 8 Mev. gammas from this reaction (5). These 8 Mev. gammas will have to penetrate from ½ to 5 in. of lead, and this will reduce their average energies to approximately the same as the energy of the radium gammas.

An experiment was performed to determine the probable error induced by the arrangement for positioning the detector holder over the neutron source. In this experiment all variables were held constant, and the detector holder was removed and replaced between readings. The probable error was found to be approximately 80 counts in 20,000, which is approximately 0.4 per cent.

The operating characteristics of the B^{10} lined neutron counter led to the conclusions that it was excellent for the detection of slow neutrons in this investigation. The one source of error which was present due to the use of this detector was the possibility of slow neutrons leaking through the cadmium inside the detector holder. This source of error could be avoided by more careful design and construction of this cadmium shield, or by performing the investigation in an area far removed from any scattering materials other than those in the shield. The B^{10} lined neutron counter is not easily adaptable for the measurement of fast neutrons. The method used in this investigation, although it was considered adequate here, is not recommended as a standard procedure to be used for future investigations.

B. Operating Characteristics of the Geiger-Müller Tube and the General Electric Radiation Monitor

An experiment was performed to determine whether or not the Geiger-Müller tube and the General Electric Radiation Monitor were sensitive to neutrons. This experiment consisted of placing the detectors in a fixed position with a ½-in. sheet of lead between them and the source of radiation. The purpose of this sheet of lead was to remove all radiation except gammas and neutrons. The sources of radiation used were the 94.7-mg. radium-beryllium neutron source and a 10.02-mg. needle of pure radium. Readings were taken with each of these sources in position with and without 4½ in. of paraffin between the source and detectors. In every case the pure radium source gave a higher reading when the observed reading were converted to a basis of one mg. of radium. This experiment indicated that both the Geiger-Müller tube and the General Electric Radiation Monitor were insensitive to both fast and slow neutrons.

The General Electric Radiation Monitor had a listed calibration of ±10 per cent. It was believed that this calibration could not be used to give a measure of the probable error in the use of this instrument, since the probable error computed from this value would have been greater than the variation in the gamma intensity for the

different configurations of the shielding materials. An experiment was performed to determine a more useable value of the probable error. A series of readings were taken with the General Electric Radiation Monitor at a fixed distance from a 10.02-mg. needle of pure radium, and the per cent probable error was computed from these readings in the following manner.

$$\text{Standard Deviation } (\sigma_x) = \sqrt{\frac{\sum_{i=1}^{N} (X_i - \bar{X})^2}{N}}$$

$$\text{Probable Error} = 0.674\,\sigma_x$$

where:

$\bar{X}$ is the average value of the individual readings.

X_i is the value of the individual readings.

N is the number of readings.

$$\text{Per Cent Probable Error} = \frac{\text{Probable Error}}{\bar{X}} \times 100.$$

The per cent probable error as defined above was computed for this series of readings and its value was determined to be 1.44. The probable error listed in Table 5 includes both this error and the assumed 0.1 mr. error in reading the scale.

Although the operating characteristics of these detectors indicated that they should have been excellent for the detection of gamma radiation in a combined field of gammas and neutrons, there were several limitations observed in

their use during this investigation. The Geiger-Müller tube counts only a very small fraction of the incident gammas, but the most severe limitation of the use of this detector is the conversion of the counts recorded into other units normally used as the measure of the intensity of gamma radiation. However, the Geiger-Müller tube is an effective instrument for making comparative measurements of the number of gammas which penetrate different configurations of a shield. The General Electric Radiation Monitor is not as accurate as some of the other ionization-chamber type instruments, but its full scale deflection of twenty milliroentgens made it desirable in this investigation. This instrument is not recommended for the measurement of the intensity of gamma radiation unless the intensity is so low that the use of other types of instruments would require unreasonably long irradiation times.

C. Scattering Losses

An experiment was performed in an attempt to determine the effects of the degree of lamination on the scattering of neutrons through the edges of the shielding materials. In this experiment the cadmium was removed from the different configurations of the shield, and the count due to neutrons of all energies was observed. Since the fast neutrons undergo fewer scattering collisions than the slow neutrons, and

since the neutron counter is much less efficient for fast neutrons than for slow neutrons, it is logical to assume that the observed scattering losses are primarily due to slow neutrons.

The counts observed in this experiment are listed in Table 6. If it could be assumed that the moderating ability

Table 6

Neutron Counts Without Cadmium In the Shield

Cycles of Shielding Materials	Counting Time (Minutes)	Counts	Counting Rate (R) $R = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}}$
1	10	27288	2728.8 ± 16.5
2	10	28160	2816.0 ± 16.8
5	10	26478	2647.8 ± 16.3
10	10	24242	2424.2 ± 15.6

of the shielding materials was not a function of the degree of lamination, the percentage increase in scattering losses between the one-cycle configuration and the ten-cycle configuration could be computed by:

$$\text{Percentage Increase in Scattering Losses} = \frac{(\text{One-cycle Count} - \text{Ten-cycle Count})}{\text{One-cycle Count}} \times 100.$$

The increase in scattering losses computed by this method was found to be 11.1 per cent.

Since it has been established that there was an increase in the moderating ability of the materials as a function of the degree of lamination, this observed value of the percentage increase in scattering losses is somewhat lower than the actual value. Due to the shape of the neutron cross section curves (Figure 10), it is logical to assume that this increase in the moderating ability of the materials is relatively small, and consequently the increase of the actual over the computed value of the percentage increase in scattering losses will be small.

D. Comparison of the Ten-cycle Configuration With a Homogeneous Shield

This experiment was performed by observing the total neutron counts after each cycle was added to the ten-cycle configuration of the shield. The observed data are listed in Table 7.

The value for the effective μ was determined by use of the equations:

$$I_1 = I_o e^{-\mu X_1} \quad \text{and} \quad I_2 = I_o e^{-\mu X_2}$$

where:

I_1 is the counts after one cycle.

I_2 is the counts after two cycles.

I_o is the counts without any shielding materials.

Table 7

Neutron Counts

Cycles of a Ten-cycle Shield	Counting Time (Minutes)	Counts	Counting Rate (R) $R = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}}$
1	3	18624	6208.0 ± 45.5
2	3	15540	5180.0 ± 41.6
3	3	13148	4383.0 ± 38.3
4	4	13894	3496.0 ± 29.5
5	5	13086	2617.0 ± 23.0
6	6	11493	1915.5 ± 18.0
7	8	10868	1458.0 ± 13.0
8	13	11696	899.7 ± 8.3
9	16	10802	676.0 ± 6.5
10	40	16008	400.2 ± 3.2

X_1 is the thickness of one cycle of materials.

X_2 is the thickness of two cycles of materials.

μ is the effective absorption coefficient for the materials.

This method was used to determine the value for the effective absorption coefficient for each cycle of shielding materials. The average value of these absorption coefficients was determined and used in the above equations to compute values of I_0. The average value of I_0 was determined and a theoretical plot was made, using the average value of I_0 and the average value of the absorption coefficient in the above equations (Figure 14). The observed counting rate was then plotted on the same graph, and it can be seen that there is

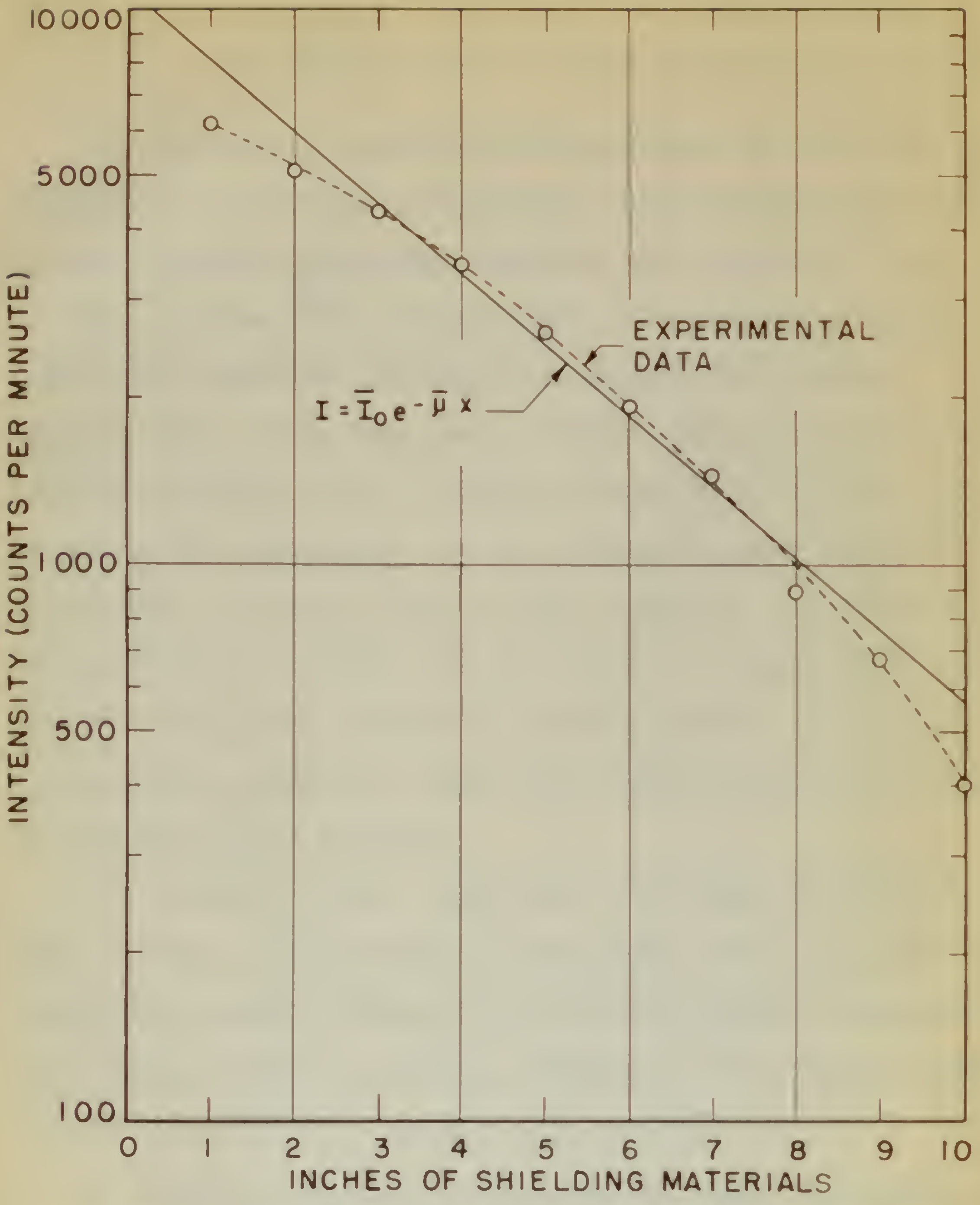

Figure 14. Shielding Characteristics of the Ten-cycle Shield.

fair agreement between the two plots, with the observed data diverging from the theoretical plot at each end.

E. Supplementary Neutron Counts

An additional experiment was performed to determine the effects of the neutrons which were scattered into the shield by the paraffin surrounding the lead ball shown in Figure 7. In this experiment a 0.04-in. sheet of cadmium was placed around the shielding materials to capture the thermal neutrons that would have been scattered into the shield from the upper surface of the paraffin surrounding the lead ball in the source container. As an additional precaution, 2½ in. of paraffin was placed next to this cadmium. The purpose of this additional paraffin was to increase the probability that neutrons which were scattered into the cadmium surrounding the shielding materials would have thermal energies and would be captured by the cadmium.

The results of this experiment are listed in Table 8. A check was made to determine if the basic system used for this experiment was the same as it was in the previous experiment. This check was made by taking readings of the neutron counts without the additional cadmium and paraffin around the edges of the shielding materials, which was the same arrangement as arrangement 1 shown in Figure 7. These results indicated that

Table 8

Neutron Count -- Corrected

Cycles of Shielding Materials	Counting Time (Minutes)	Counts	Counting Rate (R) $R = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}}$
1	30	11650	388.3 ± 3.6
2	30	11421	380.7 ± 3.6
5	30	11994	399.8 ± 3.6
10	30	11818	393.9 ± 3.6

the two systems were similar but were not identical.

No experiments were performed to determine the effects of these scattered neutrons on the fast neutron count. Since there is a finite probability that neutrons of all energies will be scattered into the shield through the edges of the shielding materials, it is logical to assume that these scattered neutrons will have some effect on the fast neutron counting rate. Since the two systems were not identical, and since the effect of the scattered neutrons on the fast neutron count was not determined, no attempt was made to separate the total count into counts due to slow neutrons and counts due to fast neutrons. The results of this experiment are plotted in Figure 15.

A second additional experiment was performed to determine the effects of the degree of lamination on the scattering of

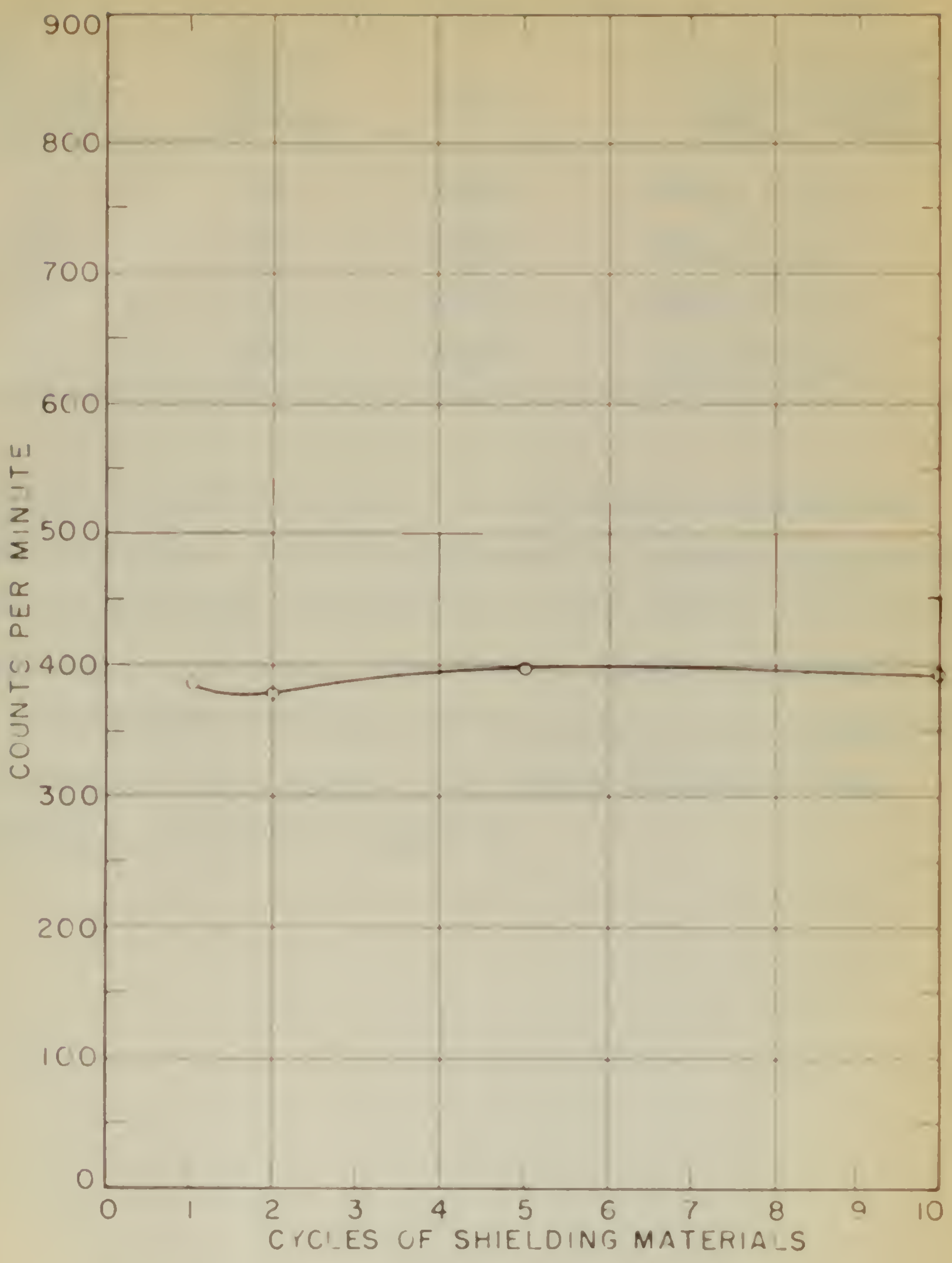

Figure 15. Supplementary neutron count.

Table 9

Corrected Neutron Counts Without Cadmium in the Shield

Cycles of Shielding Materials	Counting Time (Minutes)	Counts	Counting Rate (R) $R = \frac{\text{Counts} \pm \sqrt{\text{Counts}}}{\text{Time in Minutes}}$
1	10	17576	1757.6 ± 13.3
2	10	22072	2207.2 ± 14.8
5	11	23862	2169.3 ± 14.0
10	10	21536	2153.6 ± 14.7

neutrons through the edges of the shielding materials for this new system. In this experiment the cadmium was removed from the different configurations of the shield, the cadmium sheet and the additional paraffin was placed around the edges of the shielding materials, and the counts due to neutrons of all energies were observed. The counts observed in this experiment are listed in Table 9.

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